MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

SUBJECT: RES REPORT "FEEDBACK FROM RECENT OPERATIONAL EXPERIENCE"

The purpose of this memorandum is to transmit the Office of Nuclear Regulatory Research (RES) report "Feedback From Recent Operational Experience" for your information. The Commission Staff Requirements Memorandum (SRM) for SECY-98-228 directed the transfer of the independent analysis and evaluation of operational data function from the former Office for Analysis and Evaluation of Operational Data (AEOD) to the Office of Nuclear Regulatory Research.

This report provides a summary update of recent operating experience analyses for use by internal and external stakeholders. The analyses in this report were published separately, but are summarized in this report to provide a consolidated, ready reference for stakeholders. Our current plans are to update this report annually. The report is responsive to concerns raised by external stakeholders that summary information previously available in the AEOD annual reports (NUREG 1272 series) was no longer available.

Information in this report was previously transmitted to the Office of Nuclear Reactor Regulation (NRR) and the Regions via memoranda (usually with attached NUREG reports). The transmittal memoranda summarized the findings and potential uses of the information in regulatory programs including the Reactor Oversight Program (ROP). For example, information contained in this report was provided to NRR and the Regions as part of the End of Cycle (EOC) review meetings in preparation for the annual Agency Action Review Meeting (AARM). In addition, this information was provided as input to the SECY paper on Industry Trending that NRR prepared in support of the annual performance report to Congress and in support of future improvements to the ROP.

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FEEDBACK FROM RECENT OPERATIONAL EXPERIENCE

FY 2000 REPORT ON REACTORS

Office of Nuclear Regulatory Research

U.S. Nuclear Regulatory Commission

June 2001

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FEEDBACK FROM RECENT OPERATIONAL EXPERIENCE

FY 2000 REPORT ON REACTORS

Background

In the Staff Requirements Memorandum for SECY-98-228, “Proposed Streamlining and Consolidation of AEOD Functions and Responsibilities,” the Commission approved the staff’s plan to streamline the Office for Analysis and Evaluation of Operational Data (AEOD) and consolidate its functions in other program offices. The Commission further directed that “[t]he lessons learned from the independent assessment of operational events must continue to be shared with the nuclear industry in an effort to improve the safety of licensed operations and to assess the effectiveness of agency wide programs,” and “[t]he Office of Research should provide focused analysis of the operational data...” In 1999, the United States (U.S.) Nuclear Regulatory Commission (NRC) reorganization was implemented and the AEOD function of evaluating operational experience and performance trends of nuclear reactors was placed in the Office of Nuclear Regulatory Research (RES). Since 1985, AEOD published annual reports presenting the staff’s overview of the operating experience of U.S. nuclear power reactors. The last such report was published in November, 1998, as NUREG-1272, Vol.11, No.1.

Message from the Director of Office of Nuclear Regulatory Research

In response to Commission direction in the SRM for SECY-98-228 to share lessons learned from the RES independent assessment of nuclear reactor operational experience with stakeholders, RES is issuing its first annual report of feedback from recent operational experience of U.S. nuclear power reactors. The purpose of this report is to provide a summary of recent operational experience for use by internal and external stakeholders. Figure 1 displays the general process for developing the annual report along with the current and/or potential uses of the information by internal stakeholders.

The concept of an "independent" review of operating experience was proposed in the Kemeney and Rogovin reports following the accident at Three Mile Island Unit 2. In this sense, "independent" analyses relates to the fact that RES (formerly AEOD) determines the risk important areas for analysis and conducts evaluations of the operating experience. However, none of these analyses are completely "independent." RES relies on data from industry [Licensee Event Reports (LERs), Equipment Performance and Information Exchange (EPIX), and Nuclear Plant Reliability Data System (NPRDS)] and internal inspection reports as the basis for the analyses. In addition, we rely on critical review and feedback from internal and external stakeholders to ensure the quality of our "independent" analyses. We then provide the results of our evaluations to the public and to the responsible program offices (NRR and the Regions) for their information and use.

Information in this report was previously transmitted to the Office of Nuclear Reactor Regulation (NRR) and the Regions via memoranda (usually with attached NUREG reports). The transmittal memoranda summarized the findings and potential uses of the information in regulatory programs including the Reactor Oversight Program (ROP). For example, information contained in this report was provided to NRR and the Regions as part of the End of Cycle (EOC) review meetings in preparation for the annual Agency Action Review Meeting (AARM). In addition, this information was provided as input to the SECY paper on Industry Trending that NRR prepared in support of the annual performance report to Congress and in support of future improvements to the ROP.

1. EXECUTIVE SUMMARY

RES performs evaluations of nuclear power plant operational events to identify potential accident sequence precursors and performance trends. The data from operational events is used to develop important operational insights and to minimize safety vulnerabilities. For example, the accident sequence precursor analysis of conditions for D.C. Cook showed that the potential high-energy line break was an important risk contributor affecting mitigative system operability. This vulnerability was addressed through the IMC-0350 process (Oversight of Reactor Facilities in a Shutdown Condition with Performance Problems) before the plant was allowed to restart. For FYs 1999 and 2000, the number of risk-important potential precursors (conditional core damage probability [CCDP]$\geq 1.0\times10^{-4}$) were one and three, respectively. During the same period, there was no event identified as a significant precursor (CCDP$\geq 1.0\times10^{-3}$) a nuclear accident. This satisfied the measure of maintaining safety as described in NRC's Strategic Plan. The occurrence rate of all precursors has exhibited a statistically significant decreasing trend during the FY1993-1999 period by about a factor of three.

The annual industry average of the seven key performance indicators from NUREG-1187 series for FY 1988 through FY 2000 are presented in this report. The number of automatic reactor scrams, safety system actuations, significant events, equipment forced outages, and collective radiation exposure showed decreasing (improving) trends during that period. The number of safety system failures and forced outage rate did not exhibit a trend during that period.

Results from four system and component reliability studies are summarized in this report. They include the high pressure safety injection (HPI) system in PWRs, turbine-driven pumps (TDPs), motor-driven pumps (MDPs), and air-operated valves (AOVs). The HPI study concluded that, from1987 through1997, there is an industry wide decreasing (improving) trend for both the unplanned demand frequency of HPI actuation (i.e., less challenge) and the frequency of reportable failure events of the HPI system (i.e., fewer failures). The study on TDPs showed that the mean probability of a TDP failure was approximately between 1.6E-2 and 3.3E-2 from 1987 through1998. The results also showed that pump subcomponent failures were usually insignificant contributors to the TDP system failure while both turbine and governor subcomponent failures were significant contributors. Furthermore, results of the study revealed no evidence of increasingly higher failure rates for any of the plant age groups, and age/wear mechanisms were not the predominant cause of TDP failure. For the MDP study, results showed that the mean probability of a MDP failure was between 1.2E-3 and 3.5E-3 from 1987 through1998, with the exception of the BWR reactor building closed cooling water system
(3.5E-4) and the high pressure core spray system (1.2E-2). The results also showed that circuit breaker failures were significant contributors to the MDP system failure in both PWR and BWR systems. Furthermore, results of the study revealed no evidence of higher failure rates as a function of plant age groups. Failure of MDP assemblies in PWRs were mainly attributed to unknown causes (root cause analysis was generally not performed for circuit breaker failures) while age/wear was the predominant cause in MDP failures in BWRs. The AOV performance study could not develop failure

trend or mechanism for BWR AOVs because of sparsity of data. For PWR AOVs, age/wear was the predominant cause of failures.

Consistent with the NRC's reactor performance goals of maintaining safety and making NRC activities and decisions more effective, efficient, and realistic, staff assessed two regulations to determine their effectiveness. This report documented the results of the regulatory effectiveness reviews for the station blackout (SBO) rule and the anticipated transient without scram (ATWS) rule. The reviews concluded that both the SBO and the ATWS rules have been effective in reducing risk to operating plants.

The following sections present the staff's analysis and evaluation of the U.S. nuclear reactor operational experience, accident sequence precursors, performance indicators and trends. Furthermore, results from the system and components reliability studies, and the regulatory effectiveness reviews are also included.

2. IDENTIFICATION OF PRECURSORS AND PERFORMANCE INDICATORS AND TRENDS

This section discusses results from the Accident Sequence Precursor (ASP) and Performance Indicator (PI) programs. In the Nuclear Reactor Safety arena of the NRC's Strategic Plan, one of the performance goal measures is that there should be "no statistically significant adverse industry trends in safety performance." The NRC reports on this measure to Congress in March of each year as part of its Performance and Accountability Report (NUREG-1542 series). The current bases for assessing performance against this measure are trends in the industry performance indicators developed by AEOD and trends identified by the ASP program administered by RES. These indicators were among those cited as demonstrating improvements in industry safety performance that contributed to the agency's decision to revise the NRC Reactor Oversight Process (ROP), which was implemented on April 2, 2000.

The results of the program and any adverse trends in industry performance indicators will be reported to Congress. The agency's response to these results will be commensurate with the safety significance of the issues. The results, along with any actions taken or planned, will be reviewed annually during the Agency Action Review Meeting (AARM) and reported to the Commission.

The indicators developed by AEOD have been published in January of each year in the NUREG-1187 series, "Performance Indicators for Operating Commercial Nuclear Power Plants," the AEOD Annual Report (NUREG-1272 series - no longer published), the NRC Annual Report (NUREG-1145 series - no longer published), and the NRC's Information Digest (NUREG-1350 series) published in the summer/fall of each year. The results of the ASP program have been published in the NUREG/CR-4674 series, "Precursors to Potential Severe Core Damage Accidents," and reported annually to the Commission since 1994, most recently in SECY-01-0034, "Status Report on Accident Sequence Precursor Program and Related Initiatives."

The industry indicators in NUREG-1187[1] are to be distinguished from the plant-specific performance indicators developed for the ROP, which are used to assess the safety performance of individual plants. No decision has been made on whether these will be phased out. Because the program has

[1] Two of the seven NUREG-1187 PIs have counterparts in the ROP: unplanned scrams and safety system failures. The other ROP indicators are substantially different from the NUREG-1187 PIs.

been transferred to the Office of Nuclear Reactor Regulation (NRR), future editions of this annual report will not include NUREG-1187 PI trends.

The ASP Program provides a measure of the overall significance of events within the reactor safety arena, and the analyses generally relate to the initiating events or mitigating systems cornerstones of the ROP. RES intends to continue its assessment of operational events using the ASP program, as well as updates to analyses of operating experience at the system, component, and event levels, and plans to continue to publish this assessment as part of the annual report.

2.1 Accident Precursor Program for FY 1999 and FY 2000

Established by the NRC in 1979 in response to the Risk Assessment Review Group report (NUREG/CR-0400, September 1978), the primary objective of the Accident Sequence Precursor (ASP) Program is to systematically evaluate U. S. nuclear plant operating experience to identify, document, and rank those operating events that were most significant in terms of the potential for inadequate core cooling and core damage (precursors). In addition, the ASP Program has the following secondary objectives: (1) to categorize the precursors for plant-specific and generic implications, (2) to provide a measure that can be used to trend nuclear plant core damage risk, and (3) to provide a partial check on PRA-predicted dominant core damage scenarios.

The program is also used to monitor the agency's performance against the following Strategic Plan goal for maintaining safety: "No more than one event per year which is a significant precursor of a nuclear reactor accident." Since its inception, the staff has published seventeen ASP Program reports documenting the results of its review of operational experience for precursors covering the years 1969-1998. These reports have been issued on a yearly basis since 1986 as the NUREG/CR-4674 series (Precursors to Potential Severe Core Damage Accidents).

Accident sequences of interest to the ASP Program are those that would have resulted in inadequate core cooling that could have caused severe core damage, if additional failures had occurred. Events or conditions considered to be potential precursors are analyzed, and a conditional core damage probability (CCDP) is calculated by mapping failures observed during the event onto accident sequences in risk models.

The staff uses the ASP methodology, models, and results as follows:

(6) To make prompt assessments of the risk significance of operational events to support regulatory decisions by senior management.

(7) For Phase 3 of the significance determination process (SDP) to evaluate the significance of inspection findings as part of the agency's improved reactor oversight process.

(8) To evaluate the change in risk associated with licensing amendments submitted by licensees requesting changes in surveillance frequencies or allowed outage times.

(9) To determine the need for generic communications (such as information notices).

(10) To conduct systematic screening, review, and analysis of operational experience data for accident sequence precursors.

(11) To evaluate the generic implications of precursors, trend industry performance, and check against PRAs.

(12) To perform regulatory analyses in association with the resolution of generic issues.

(13) To evaluate the risk associated with a specific technical issue identified at an individual plant.

2.2 FY 1999 and FY 2000 Results[2]

The results of final and preliminary ASP analyses for FYs 1999 and 2000 are presented in Tables 2.1 through 2.4. All of the precursors during this time were at-power events; there were no shutdown precursors.

Important precursors. Precursors with a CCDP (for initiating events) or importance (for equipment unavailabilities) $\geq 1.0\times10^{-4}$ are considered important with respect to risk significance. The number of potential important precursors that occurred in FYs 1999 and 2000 are one and three, respectively. The average number of important precursors per year during the 1993-1998 period was two.

Two of the potential precursors that occurred in FYs 1999 and 2000 involved conditions that could render safety systems inoperable during postulated high-energy line breaks at both units of D. C. Cook. The two other potential precursors involved an extended loss of offsite power at Diablo Canyon 1 and at Indian Point 2.

Significant precursors. The ASP Program is used to monitor the agency's performance against the following Strategic Plan performance goal measure: "No more than one event per year identified as a

[2]Current as of March 30, 2001

significant precursor of a nuclear accident." A "significant precursor" is defined in the Plan as such events that have a 1/1000 (10^{-3}) or greater probability of leading to a reactor accident.

No potential precursor was identified during FYs 1999 and 2000 with a CCDP$\geq 1.0\times10^{-3}$. Precursors with CCDP$\geq 1.0\times10^{-3}$ have occurred, on the average, about once every 4 years. The events in this group appear to exhibit no common (generic) characteristics with respect to the nature, modes, causes, and systems affected by the events.

Two precursors with a CCDP$\geq 1.0\times10^{-3}$ have occurred since 1991–the Wolf Creek reactor coolant system draindown to the refueling water storage tank during hot shutdown (1994) and the Catawba 2 extended plant-centered loss of offsite power with an emergency diesel generator out of service for maintenance (1996).

2.3 FY 1993-1999 ASP Trends

Trends of the FY 1993-1999 ASP data are provided below.

- The occurrence rate of precursors has exhibited a statistically significant decreasing trend during the 1993-1999 period (Figure 2.1). The number of precursors has decreased over the period by about a factor of three.

- The occurrence rate for the potential precursors in FY 1999 is consistent with the end point of this trend, as shown in Figure 2.1.

- A chart showing CCDP "probability bins" for ASP results from FYs 1993 through 2000 is provided in Figure 2.2. The expectation of future occurrence of precursors in each ASP CCDP bin are provided below.

CCDP bin	Occurrence expectation
$\geq 10^{-3}$	1 event in 4 years
10^{-4}	3 events per year
10^{-5}	3-4 events per year
10^{-6}	3-4 events per year

- As seen from Tables 2.1 through 2.4, the total number of potential precursors for FYs 1999 and 2000 that involved the unavailability of equipment and initiating events are nine and four, respectively. These preliminary results are also consistent with the 1993-1998 results in which conditional unavailability events (63%) outnumbered initiating events (37%).

Table 2.1 FY 2000 At-Power Precursors Involving Initiating Events[3]

[3] Results as of March 30, 2001

Plant	Description/Event identifier	Plant type	Event date	CCDP	Event type
Indian Point 2	Manual reactor trip following a steam generator tube failure (LER 247/00-001)	PWR	2/15/00	8.0×10^{-5} (Preliminary)	Steam generator tube rupture
Diablo Canyon 1	Reactor trip and extended plant-centered loss of offsite power (LER 275/00-004)	PWR	5/15/00	3.1×10^{-4} (Preliminary)	Loss of offsite power

Table 2.2 FY 2000 At-Power Precursors Involving Unavailabilities[3]

Plant	Description/Event identifier	Plant type	Event date	CCDP	Importance (CCDP – CDP)	Event type
Cook 1 and 2	Potential high-energy line break conditions affecting mitigating systems operability (LER 315/99-026)	PWR	10/22/99	4.5×10^{-4}	4.3×10^{-4}	Unavail-ability
Cook 1 and 2	Valves required to operate post-accident could fail to open due to pressure locking/thermal binding (LER 315/99-031)	PWR	12/30/99	5.7×10^{-5}	3.7×10^{-5}	Unavail-ability

Table 2.3 FY 1999 At-Power Precursors Involving Initiating Events

Plant	Description/Event Identifier	Plant type	Event date	CCDP	Event type
Davis-Besse	Manual reactor trip while recovering from a component cooling system leak and de-energizing of safety-related bus D1 and nonsafety bus D2 (LER 346/98-011)	PWR	10/14/98	1.4×10^{-5}	Transient
Indian Point 2	Loss of offsite power to safety-related buses following a reactor trip and tripping of an emergency diesel output breaker (LER 247/99-015)	PWR	08/31/99	1.1×10^{-4} (Preliminary)	Loss of offsite power

Table 2.4 FY 1999 At-Power Precursors Involving Unavailabilities

Plant	Description/Event identifier	Plant type	Event date	CCDP	Importance (CCDP – CDP)	Event type
Oconee 1	Postulated high-energy line leaks or breaks in turbine building leading to failure of safety-related 4 kV switchgear (LER 269/99-001)	PWR	2/24/99	1.2×10^{-4} (Preliminary)	9.6×10^{-5} (Preliminary)	Unavail-ability
Oconee 2	Postulated high-energy line leaks or breaks in turbine building leading to failure of safety-related 4 kV switchgear (LER 269/99-001)	PWR	2/24/99	7.4×10^{-5} (Preliminary)	4.8×10^{-5} (Preliminary)	Unavail-ability
Oconee 3	Postulated high-energy line leaks or breaks in turbine building leading to failure of safety-related 4 kV switchgear (LER 269/99-001)	PWR	2/24/99	7.2×10^{-5} (Preliminary)	4.6×10^{-5} (Preliminary)	Unavail-ability
Cook 1 and 2	Lack of capability to operate emergency service water following a seismic event (Inspection reports: 50-315/316/97-024 50-315/316/99-010)	PWR	6/11/99	5.2×10^{-5}	3.2×10^{-5}	Unavail-ability

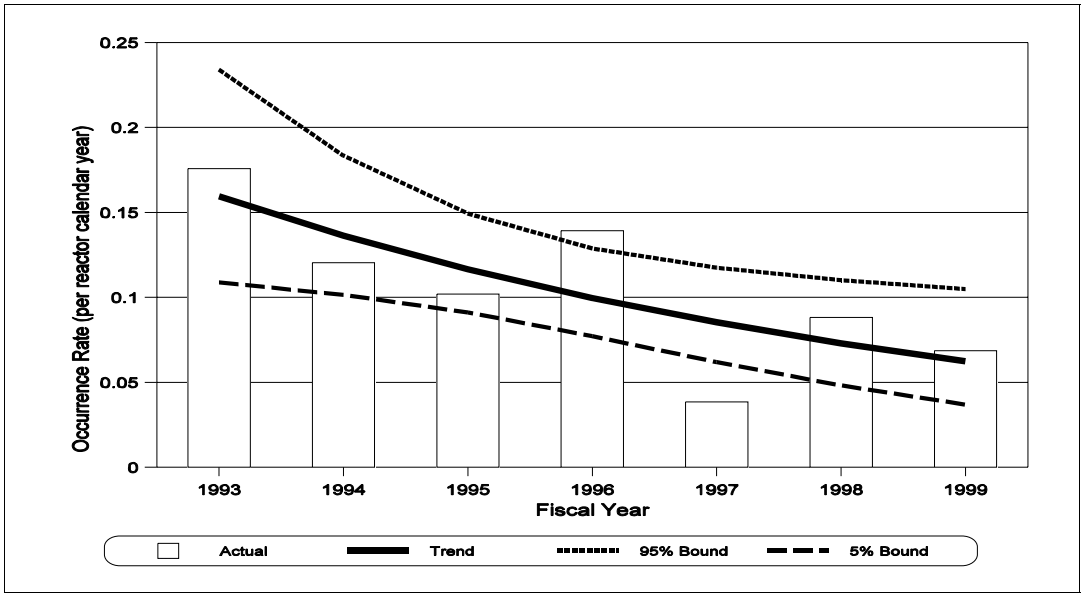

Figure 2.1 Precursor occurrence rate for 1993-1999 plotted against Fiscal Year. The trend is statistically significant (p-value = 0.0068). The result for 1999 is preliminary.

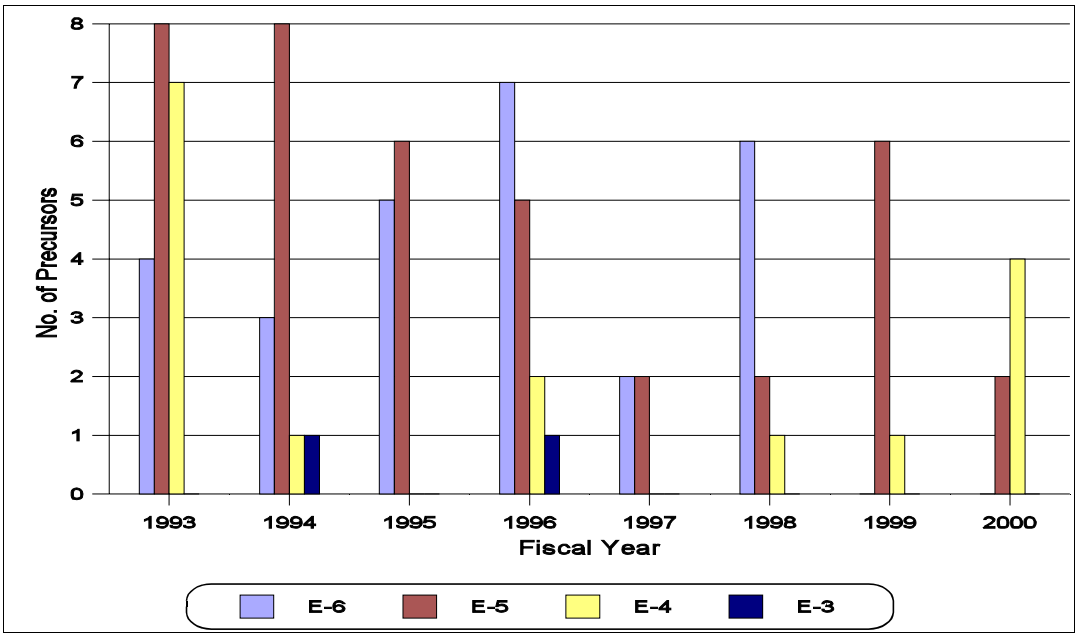

Figure 2.2 Conditional core damage probability results from ASP Program (1993-2000) for each of the CCDP bins (*E-3*: $\geq 1 \times 10^{-3}$; *E-4*: 9.9×10^{-4} to 1.0×10^{-4}; *E-5*: 9.9×10^{-5} to 1.0×10^{-5}; *E-6*: 9.9×10^{-6} to 1.0×10^{-6}). Results for FYs 1999 and 2000 are preliminary.

2.4 Performance Indicator - Annual Industry Average Trend Plots Through FY 2000

Figure 2.3 presents industry-wide annual industry averages since FY 1988 for seven of the performance indicators (PIs): automatic reactor scrams while critical, safety system actuations (SSAs), significant events (SEs), safety system failures (SSFs), forced outage rate (FOR), equipment forced outages per 1000 commercial critical hours (EFOs), and collective radiation exposure. The NUREG-1187 series (Performance Indicators for Operating Commercial Nuclear Power Reactors) provides the definitions, data sources, and computational methods used to calculate these performance indicators.

Licensees are required by 10 CFR 50.73 to report events or conditions that alone could have prevented fulfillment of various safety functions. These reports are captured to generate the performance indicator for Safety System Failures. The Safety System Failures were defined as cases where there either was or could have been a failure of a system to properly perform a safety function. Initially, it was decided that this indicator should come directly from the licensee explicitly identifying that a loss of system safety function had occurred by marking the appropriate reportability box on the licensee event report (LER) form, according to 50.73(a)(2)(v). It was later decided that this may not capture all of the safety system failures. Therefore, an improved classification procedure was implemented. Beginning with events in 1987, all LERs were screened to determine if they contained safety system failures as defined above, not just those marked by licensees. This change in coding LERs resulted in an increase in the number of safety system failures from FY 1988 on, as shown in Figure 2.3

2.4.1 Statistical Analysis of Performance Indicator Trends

As part of an assessment of PI trends, RES performed a regression analysis to evaluate the rate of change in the following PIs. This analysis is similar to an analysis performed in the 1997 AEOD Annual Report (USNRC, NUREG-1272, Vol. 11, No. 1, November 1998), and covers Fiscal Year (FY) 1988 through FY 2000.

- Automatic Reactor Scrams
- Safety System Actuations
- Significant Events
- Safety System Failures
- Forced Outage Rate
- Equipment Forced Outages
- Radiation Exposure

The results of the regression analysis are summarized in Table 2.5. Figures 2.4 through 2.10 show the seven PIs listed in Table 2.5. Scrams, SSAs, SEs, EFOs, and Radiation Exposure have statistically significant exponential model fits, indicating that a statistically significant decreasing (improving) trend is present. The SSF and FOR PIs have neither a non-linear nor a

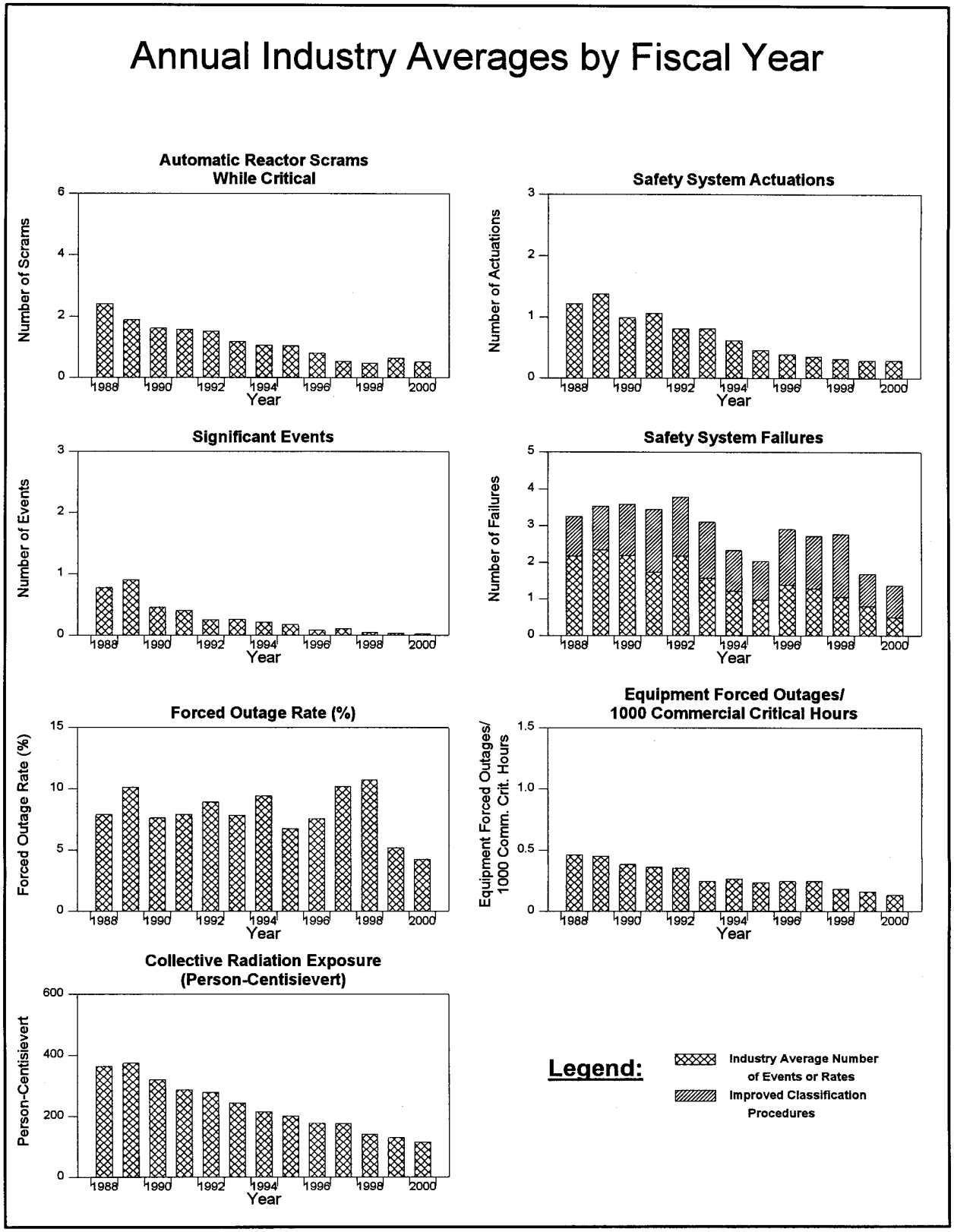

Figure 2.3 Performance Indicators - Annual Industry Averages

linear trend over the 13-year period; FOR was constant (level) over the time of interest. SSFs were modeled as having two different mean values over the periods before and after the start of FY 1994 due to implementing an improved SSF classification criterion and methodology in early FY 1994. Although the last two years of data for both SSFs and FOR appear to indicate a decreasing trend, no statistical model fit the data well enough to indicate that these last two years represent a statistically significant trend.

Table 2.5. Summary of PI Regression Analysis Trends

Performance Indicator	FY 1988-2000
Automatic reactor scrams	Statistically significant decreasing nonlinear trend
SSAs*	Statistically significant decreasing nonlinear trend
SEs	Statistically significant decreasing nonlinear trend
SSFs	Random variation over each of two periods
FOR	Random variation
EFOs	Statistically significant decreasing nonlinear trend
Radiation exposure	Statistically significant decreasing nonlinear trend

*
SSAs - Safety System Actuations
SEs - Significant events
SSFs - Safety System Failures
FOR - Forced Outage Rate
EFOs - Equipment Forced Outages

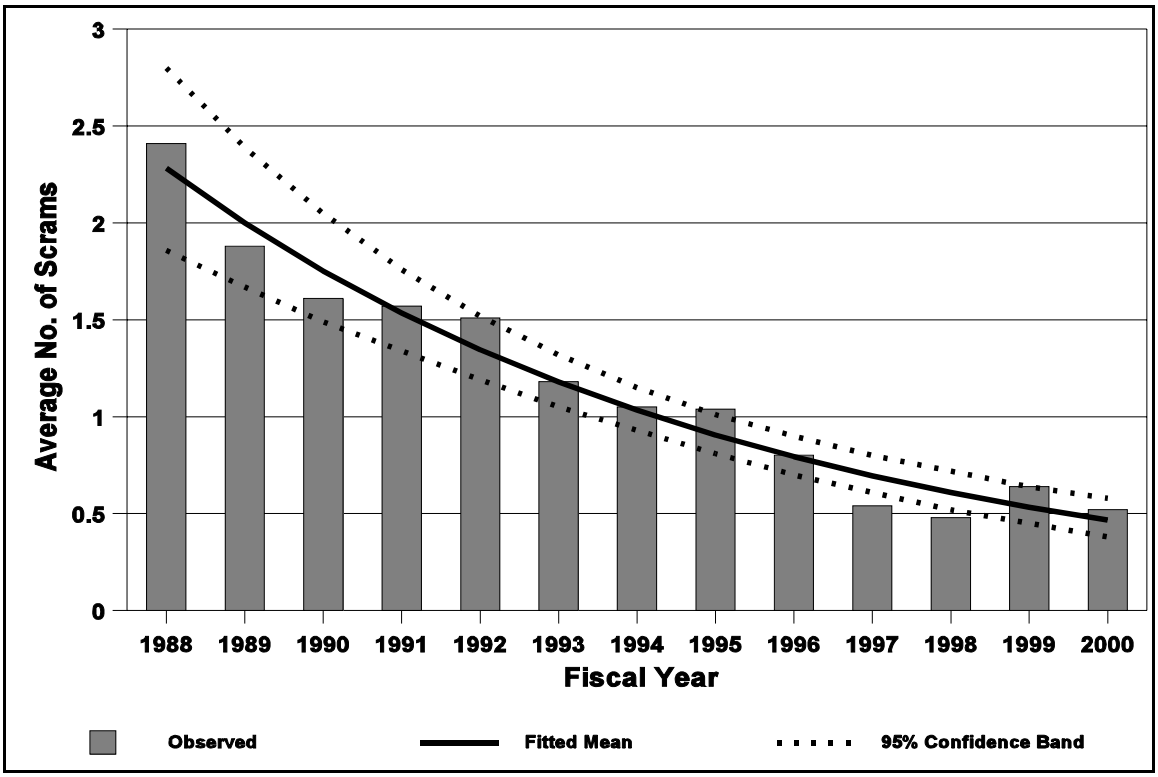

Figure 2.4 Nonlinear Regression Fit for Automatic Reactor Scrams ($y = Ae^{Bx}$)

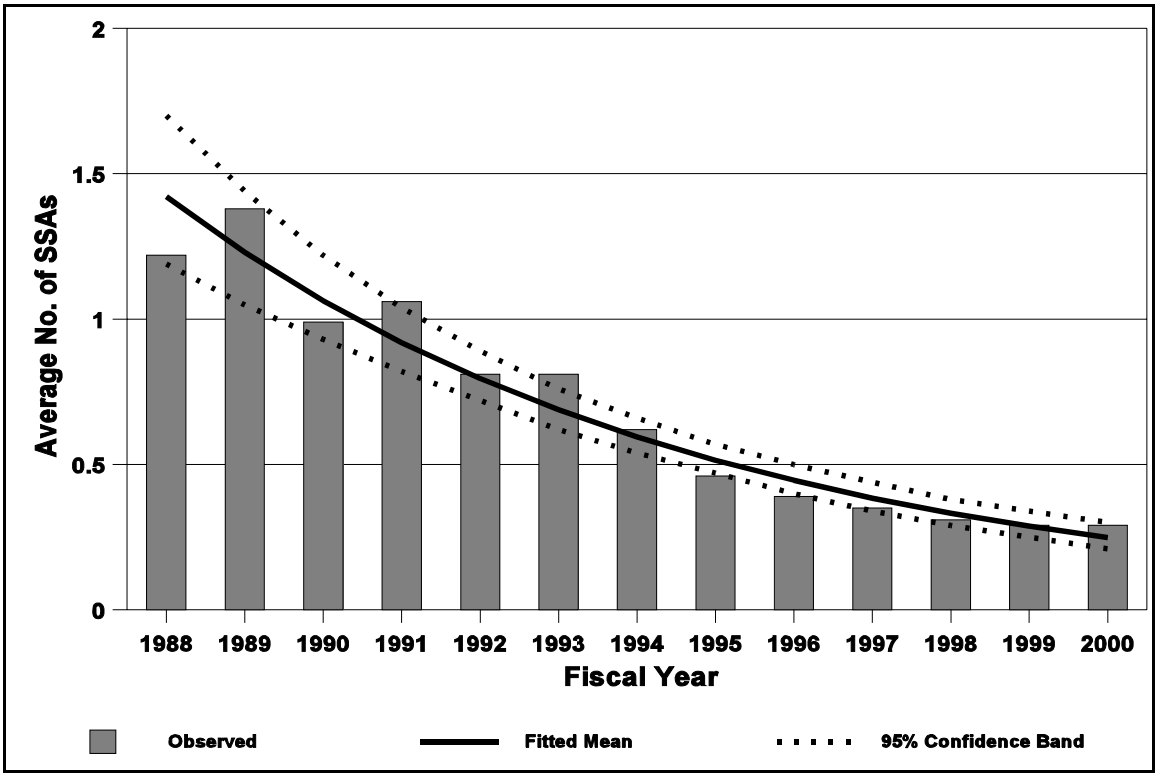

Figure 2.5 Nonlinear Regression Fit for Safety System Actuations ($y = Ae^{Bx}$)

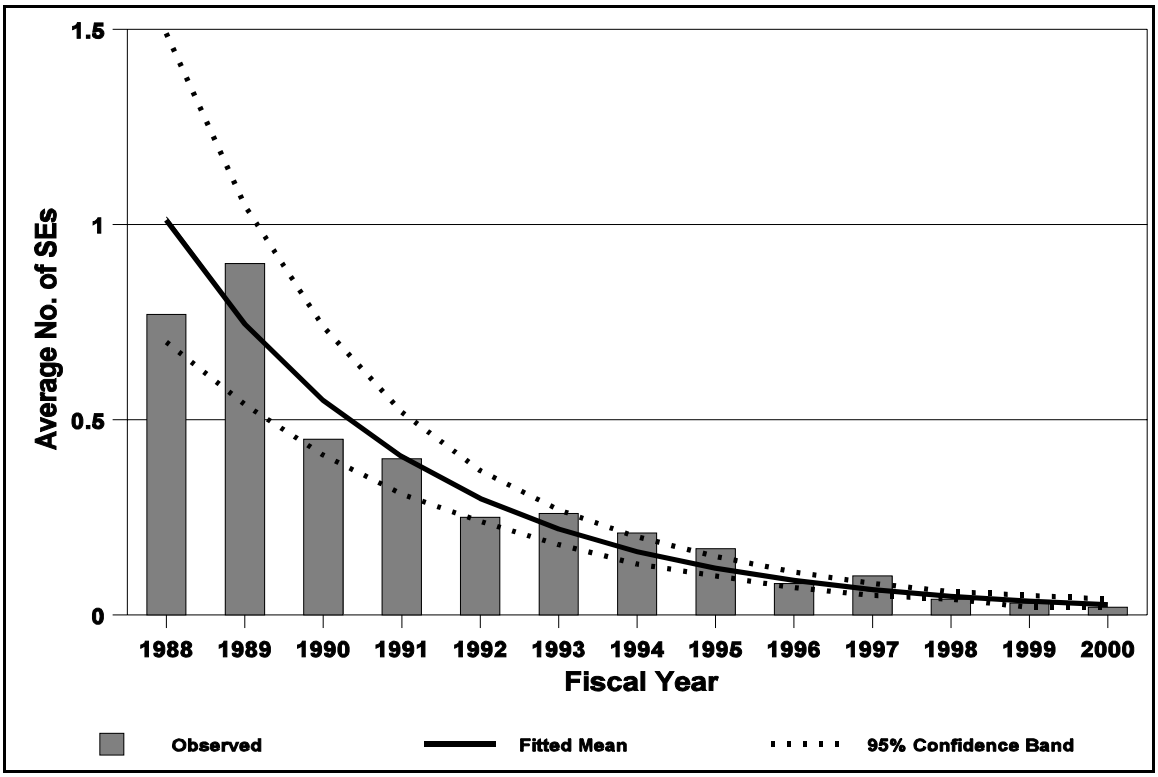

Figure 2.6 Nonlinear Regression Fit for Significant Events ($y = Ae^{Bx}$)

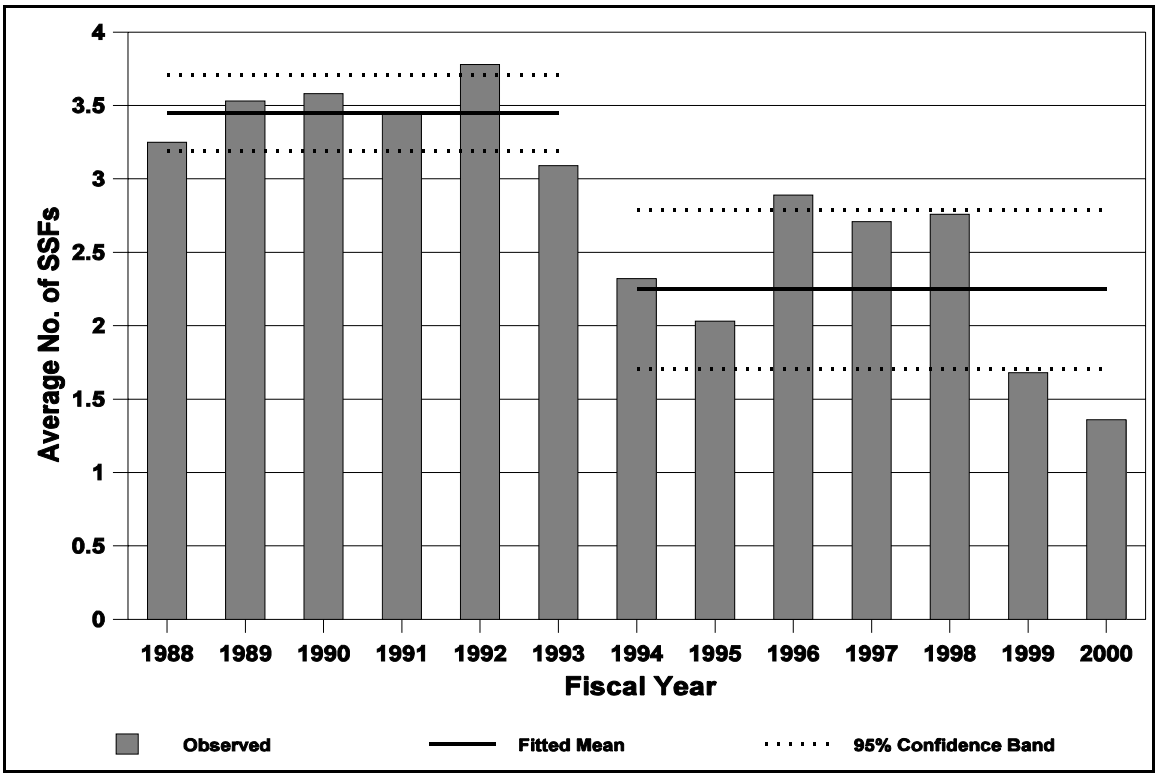

Figure 2.7 Two - Value Fit for Safety System Failures ($y = A_0$ or A_1)

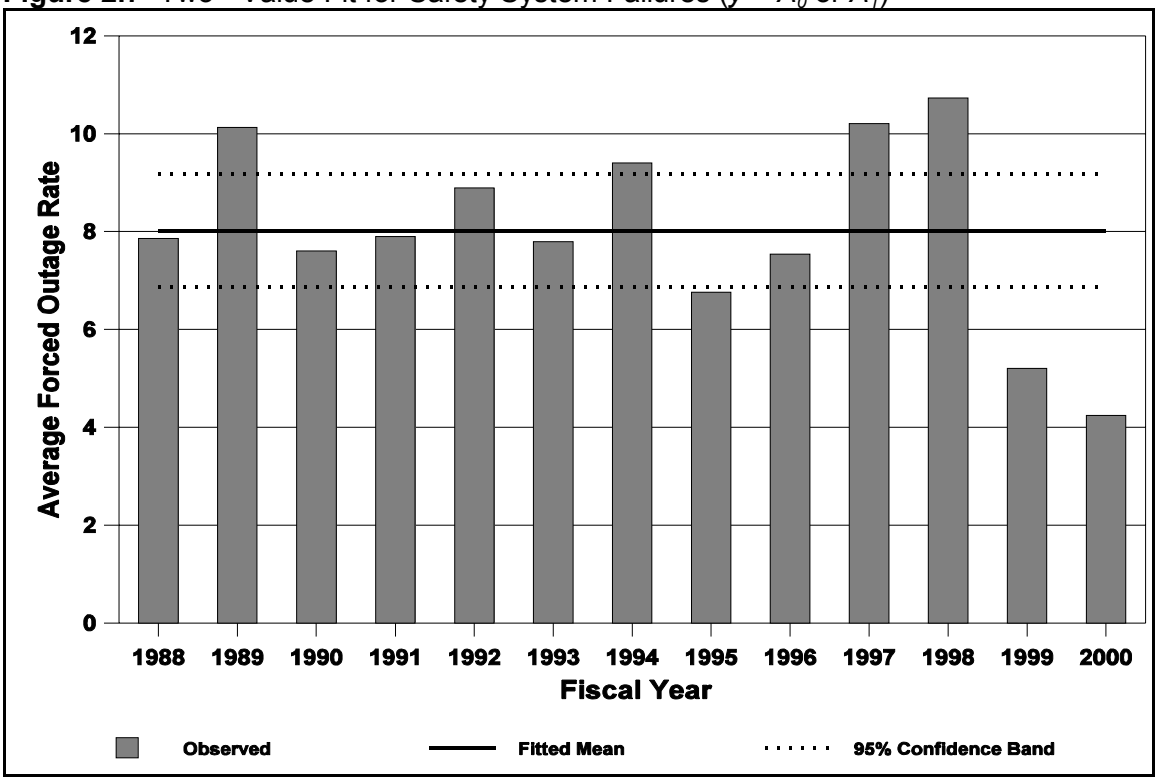

Figure 2.8 Constant Fit for Forced Outage Rate ($y = A$)

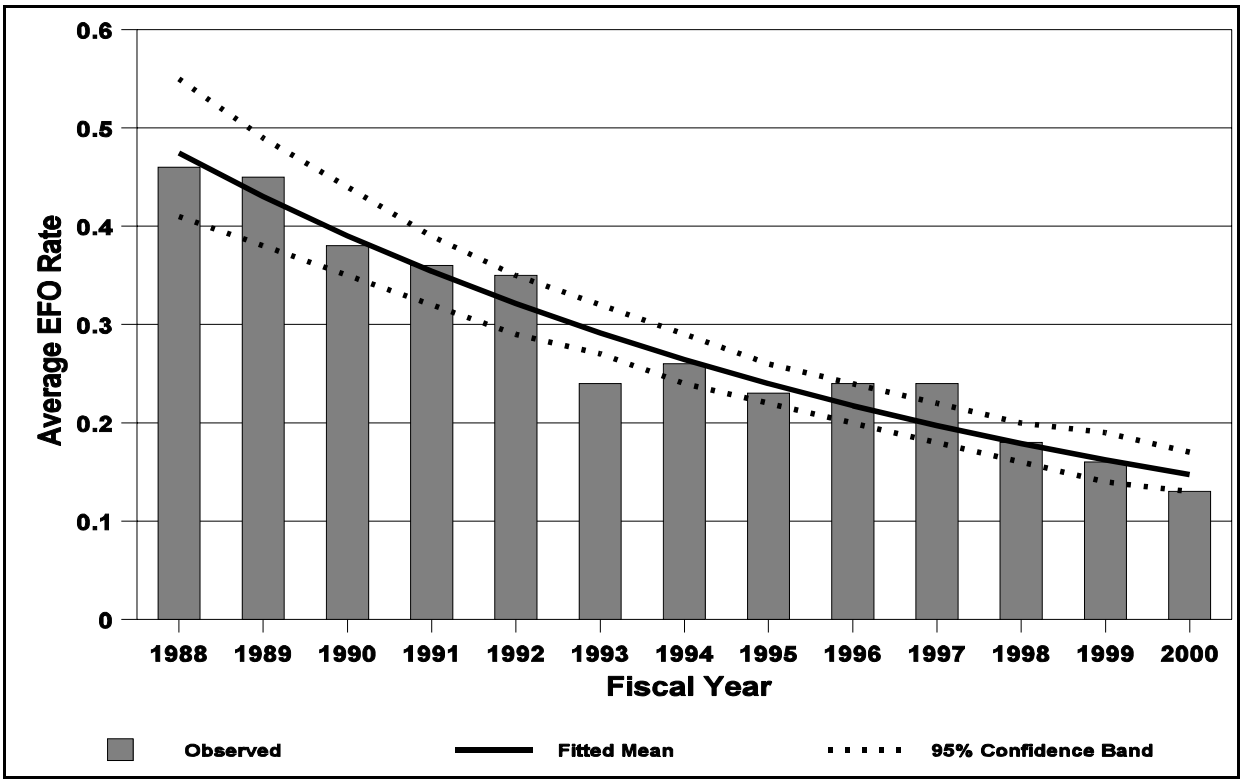

Figure 2.9 Nonlinear Regression Fit for Equipment Forced Outage Rate ($y = Ae^{Bx}$)

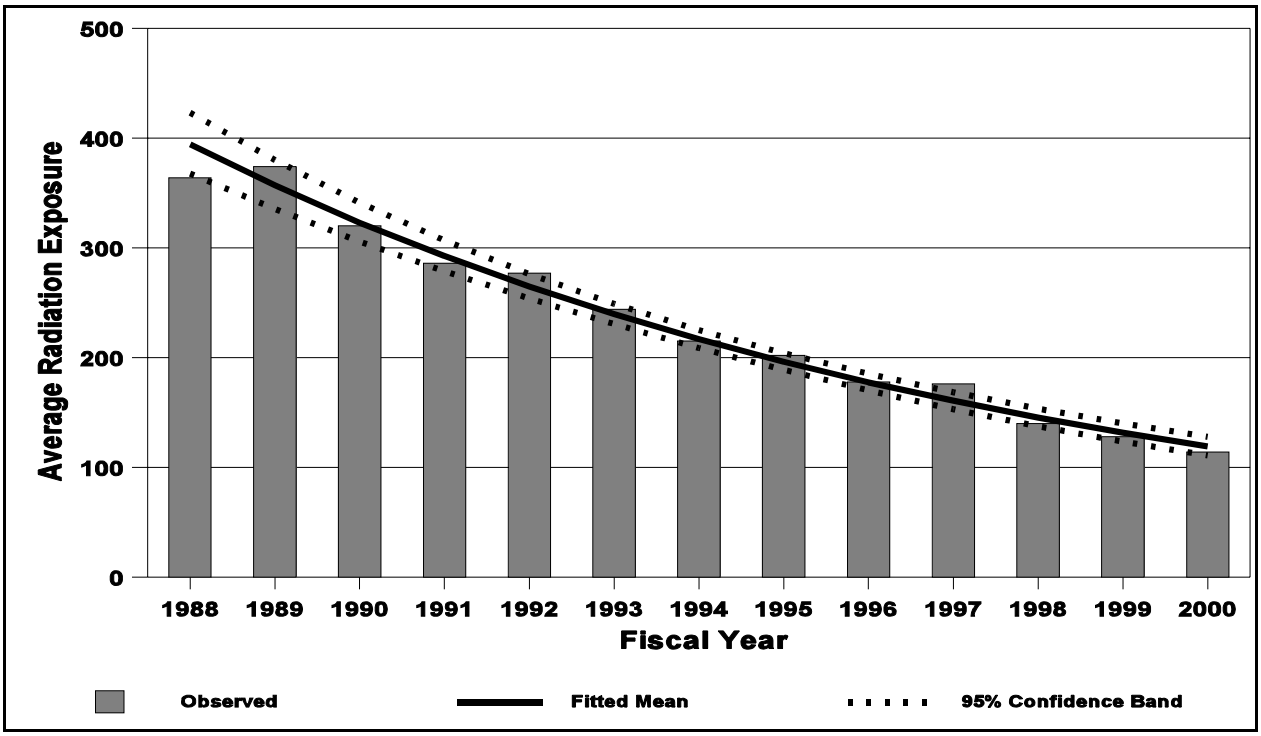

Figure 2.10 Nonlinear Regression Fit for Radiation Exposure ($y = Ae^{Bx}$)

Unplanned automatic reactor scrams while critical show a decreasing nonlinear trend that has decreased from approximately 2.5 scrams per reactor year in FY 1988 to approximately 0.5 in FY 1997 through FY 2000, as shown in Figure 2.4.

Safety system actuations show a decreasing nonlinear trend that has decreased from approximately 1.5 actuations per reactor year in FY 1988 to approximately 0.3 in FY 2000, as shown in Figure 2.5.

Significant events show a decreasing nonlinear trend that has decreased from approximately 1 event per reactor year in FY 1988 to approximately 0.02 events per reactor year in FY 2000, as shown in Figure 2.6.

Safety system failures (SSFs) have decreased since FY 1988, with some scatter in the data. From FY 1988 through FY 1993, SSFs averaged approximately 3.5 per reactor year and then averaged approximately 2.2 per reactor year since FY 1994. The number of SSFs has again decreased from FY 1998 to FY 2000. The number of SSFs per year is shown in Figure 2.7.

Forced outage rate remained fairly constant at about 8 percent from FY 1988 trough FY 1998, but has decreased somewhat over the last two fiscal years, as shown in Figure 2.8.

Equipment forced outages per 1000 commercial critical hours has decreased from approximately 0.5 in FY 1988 to about 0.15 in FY 2000, as shown in Figure 2.9.

Collective radiation exposures has decreased from approximately 370 person-rem in FY 1988 and FY 1989 to approximately 115 in FY 2000, as shown in Figure 2.10.

3. SYSTEM AND COMPONENT RELIABILITY STUDIES

The purpose of the system and component reliability studies is to evaluate reliability and to provide engineering insights of risk-important (RI) systems and components based on operating experience. The objectives of the studies are to use actual demands, failures, and unavailabilities to estimate reliability, to analyze trends in reliability, to quantify uncertainties, to compare findings with published probabilistic risk assessment and individual plant examination (PRA/IPE) values, to identify plant-specific differences, and to provide engineering insights. NRR is developing an industry trends program for operating power reactors. That program will use results and data from these studies. The current plan is that, within one year, RES will update the data and related analyses that were most recently published in NUREG-5750, “Initiating Events at U.S. Nuclear Power Plants: 1987-1995,” for use in the initiating event cornerstone. Furthermore, within 2-3 years, RES will update the data that has been published in various NUREGs for system reliability studies, component reliability studies, common-cause failure studies, and other special studies where industry-wide trends were reported, for use in the mitigating systems cornerstone and discussion in the AARM.

3.1 Reliability Study: High-Pressure Safety Injection System, 1987-1997, NUREG/CR-5500, Vol. 9

This report presents a performance analysis of High-Pressure Safety Injection (HPI) systems at United States commercial pressurized water reactors (PWRs). The evaluation is based on the operating experience from 1987 through 1997, as reported in Licensee Event Reports (LERs).

The industry-wide average for the unreliability of the HPI system for pressurized water reactors calculated from the 1987-1997 operating experience is 4.5E-4 (calculated by arithmetically averaging the results of 72 plant-specific models). Individual plant results vary by a factor of approximately 50, from 6.0E-5 to 3.5E-3. This variability is attributed to the differences in design and operating characteristics (such as number of pump trains and number of running versus standby pump trains) rather than differences in plant-specific reliability of components (such as pumps and valves).

There was no evidence of a trend in system unreliability because there were only a few failures during unplanned demands over the 11-year period covered by this report.

A statistically significant decreasing trend was identified in the frequency of unplanned demands for HPI when modeled as a function of calendar year (Figure 3-1). The unplanned demand frequency for HPI for a population of 72 PWRs decreased from approximately 48 per year in 1987 to approximately 6 per year in 1997. This constitutes an improving trend in the frequency of events challenging the HPI system.

A statistically significant decreasing trend was identified in the frequency of reportable failure events of the HPI system when modeled as a function of calendar year (Figure 3-2). The frequency of reportable failure events of the HPI system for a population of 72 PWRs decreased from approximately 18 per year in 1987 to approximately 7 per year in 1997. This trend in the reportable failure events constitutes an improving trend in the frequency of failures affecting HPI performance.

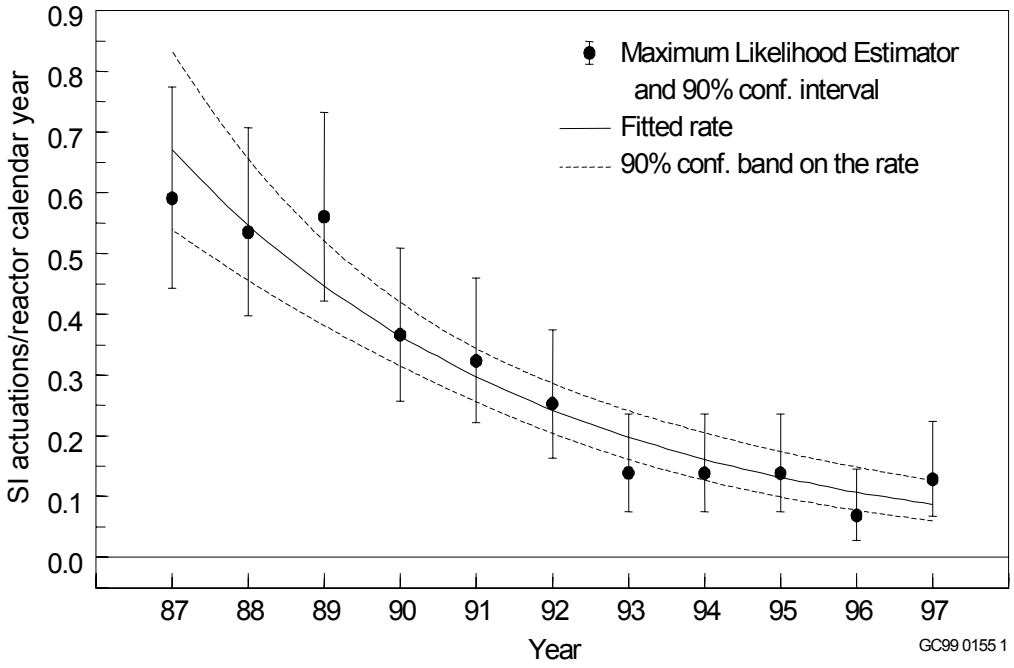

Figure 3-1. Frequency of all safety injection actuations as a function of calendar year. The decreasing trend is highly statistically significant.

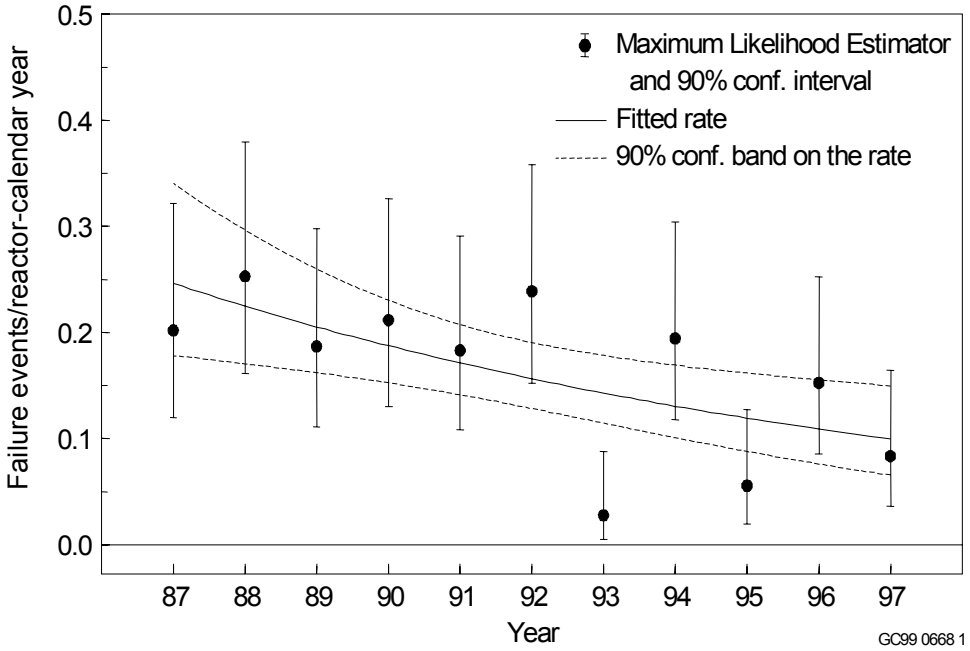

Figure 3-2. Frequency of all HPI failure events as a function of calendar year. The decreasing trend is statistically significant.

Results from additional analyses were also presented in NUREG/CR-5500, Vol. 9. For example, a statistically significant increasing trend was identified in the frequency of unplanned demands when modeled as a function of low power license date. This means that frequency of unplanned demands for HPI at plants that were licensed most recently, on the average, is expected to be higher than that of older plants. However, the absolute difference in the frequency was small. For example, based on the curve generated using operating experience from all operating plants, a plant that received the low-power license in 1967 can be expected to experience an unplanned HPI demand about once every 4 years. In comparison, again based on the curve generated using operating experience from all operating plants, a plant that received the low-power license in 1993 can be expected to experience an unplanned demand once every 2½ years. Furthermore, the significance of this difference is limited, since the frequency of total unplanned demands from all plants has been trending down yearly.

On the other hand, a statistically significant decreasing trend was identified in the frequency of reportable failure events of the HPI system when modeled as a function of low power license date. This means that frequency of reportable failure events in the HPI systems of plants that were licensed most recently, on the average, is expected to be lower than that of older plants. For example, based on the curve generated using operating experience from all operating plants, a plant that received a low-power license in 1967 can be expected to report a failure affecting HPI about once every 4 years. In comparison, again based on the curve generated using operating experience from all operating plants, a plant that received the low-power license in 1993 can be expected to report a failure affecting HPI about once every 10 years. The significance of this difference is limited, since the frequency of reportable failures from all plants has been trending down yearly. Furthermore, an examination of the nature of failures associated with older versus newer plants showed that the observed trend is not indicative of aging.

The industry-wide arithmetic average of the HPI system unreliability using data extracted from PRA/IPEs is 5.8E-4. The corresponding estimate based on operating experience is 4.5E-4. Some plants equipped with both high- and intermediate-head pumps had IPE-based unreliabilities that are more than one order of magnitude less than those unreliabilities calculated using operating experience. This difference is attributed to the low probabilities assigned to passive components and trains in the common refueling water storage tank (RWST) suction paths in these IPEs.

3.2 Component Performance Study - Turbine-Driven Pumps, 1987-1998, NUREG-1715, Vol. 1

This study provides the performance evaluation based on industry experience during the 1987 through 1998 period for turbine-driven pumps (TDPs) in the PWR auxiliary feedwater (AFW) system and in the boiling water reactor (BWR) reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems.

Table 3.1 lists the TDP probability of failure on demand estimates developed in this study for the AFW, RCIC, and HPCI systems and the generic values referenced in NUREG/CR-4550 (used as an input to NUREG-1150). For the BWR HPCI system, the probability of failure on demand over the 1987-1995 period showed a constantly decreasing trend.

TABLE 3.1
TDP PROBABILITY OF FAILURE ON DEMAND (1987-1998)

SYSTEM/SOURCE	LOWER BOUND	MEAN	UPPER BOUND
NUREG-4550	1.1E-3	3E-2	1.1E-1
AFW system	1.3E-3	1.6E-2	4.6E-2
RCIC system	9.1E-6	2.0E-2	8.7E-2
HPCI system	1.6E-3	3.3E-2	9.7E-2

The TDP mean probabilities of failure on demand used in plant-specific IPE studies were compared with the results of this study. For BWR RCIC and HPCI systems (1987-1995 data), all of the IPE mean values for the TDP failure on demand probability were within the range of this study and NUREG/CR-4550. For the AFW system, more than 90 percent of the IPE mean values were also within the range of this study and NUREG/CR-4550.

Failure trends for the PWR AFW system during the 1987-1995 period[4] were relatively constant, except for an upward peak in 1989 and 1990. For BWRs (RCIC and HPCI systems combined), there was a marked decreasing trend after 1991. Figure 3-3 shows the TDP failure trends for the 1987-1995 period.

[4] Due to change over from the industry's Nuclear Plant Reliability Data System (NRDS) to the Equipment Performance Information Exchange System (EPIX) there was insufficient data available for trending purposes beyond 1995 for the TDP, MDP, and AOV studies. Data on failures and demands from ESF actuations was available through the entire 1987-1998 period.

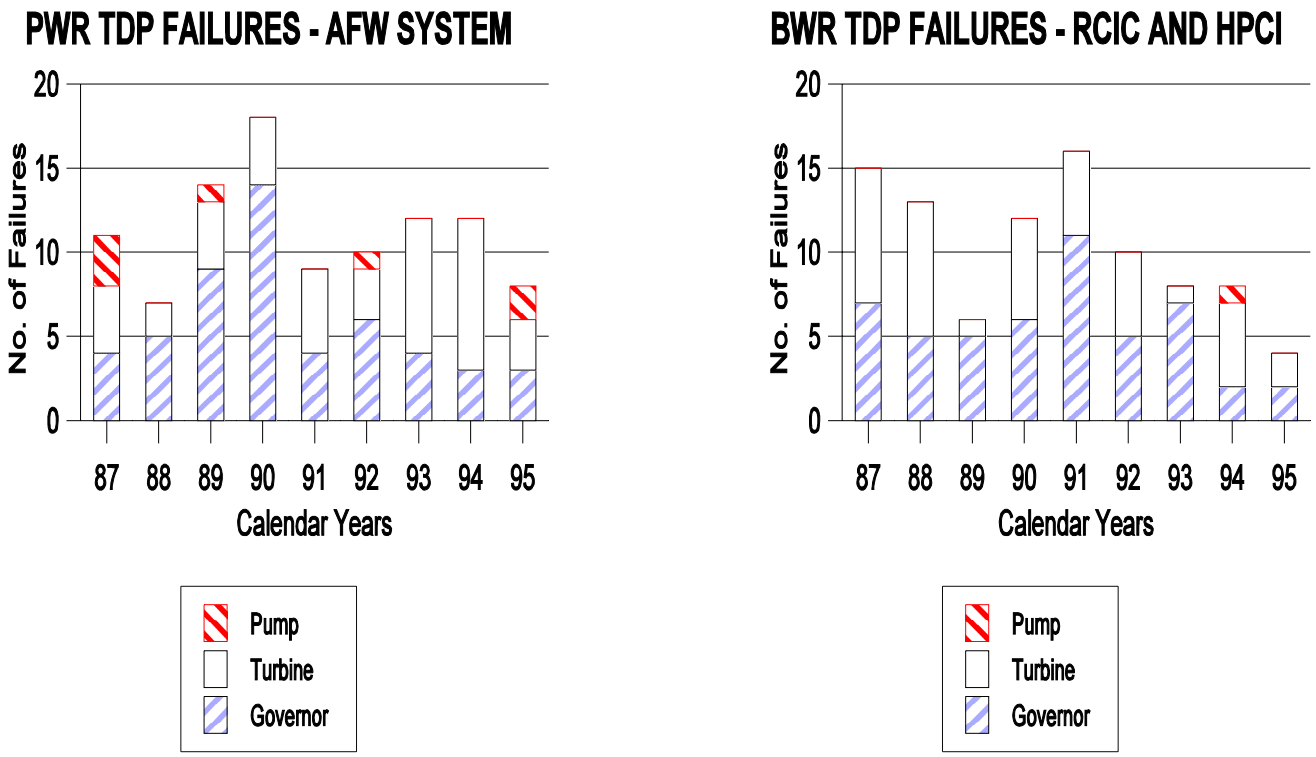

Figure 3-3
PWR and BWR TDP Failure Trends

Failure rates, as a function of component-years, were compared among the PWR and BWR plant age groups (three groups, of approximately equal size, from older to newer plants by commercial operation date). For PWRs and BWRs, the review of plant age groups did not show evidence of an increase in TDP failure rates for any of the plant age groups due to aging mechanisms.

The evaluation of TDP subcomponent failure patterns demonstrated that failures of governor subcomponents were significant contributors to the TDP failures in the BWR RCIC system, while both turbine and governor subcomponent failures were significant contributors to the PWR AFW system and BWR HPCI system. Pump subcomponent failures were relatively insignificant.

Failures of TDP assemblies in AFW and RCIC systems were mainly due both to age/wear and maintenance/procedural deficiencies, whereas maintenance/procedural deficiencies was singularly predominant for the HPCI system.

3.3 Component Performance Study - Motor-Driven Pumps, 1987-1998, NUREG-1715, Vol. 2

This study provides the performance evaluation based on industry experience during the 1987 through 1998 period for motor-driven pumps (MDPs) in the PWR and in the BWR risk important systems.

The MDP probability of failure on demand estimates were consistent with the generic values from NUREG/CR-4550 (used as an input to NUREG-1150), with two exceptions. The values for the BWR reactor building closed cooling water (RBCC) system were lower than the generic values and the mean value for the high pressure core spray (HPCS) system was higher than the generic values. Table 3.2 lists the MDP probability of failure on demand estimates developed for the RI systems selected for this study and the NUREG/CR-4550 values.

TABLE 3.2
MDP PROBABILITY OF FAILURE ON DEMAND (1987-1998)

	LOWER BOUND	MEAN	UPPER BOUND
NUREG/CR-4550	**1.1E-3**	**3E-3**	**1.1E-2**
PWR RI SYSTEMS			
Auxiliary Feedwater (AFW)	1.2E-3	1.8E-3	2.4E-3
High Pressure Injection (HPI)	9.5E-5	3.0E-3	9.6E-3
Component Cooling Water (CCW)	8.8E-7	1.4E-3	5.8E-3
Containment Spray (CS)	8.9E-5	2.1E-3	6.5E-3
Chemical and Volume Control (CVCS)	9.9E-4	2.0E-3	3.4E-3
Nuclear Service Water (NSW)	1.5E-4	2.1E-3	5.8E-3
Residual Heat Removal (RHR)	2.0E-4	1.7E-3	4.5E-3
BWR RI SYSTEMS			
High Pressure Core Spray (HPCS)	2.5E-8	1.2E-2	6.1E-2
Low Pressure Core Spray (LPCS)	2.5E-4	1.5E-3	3.6E-3
Reactor Building Closed Cooling Water (RBCC)	4.2E-5	3.5E-4	9.2E-4
Essential Service Water (ESW)	1.5E-3	3.4E-3	5.9E-3
Residual Heat Removal (RHR)	5.1E-4	1.2E-3	2.2E-3

The yearly trend analysis of the MDP probability of failure on demand showed no trend for PWR and BWR RI systems, except for the PWR HPI system. The HPI system showed an increasing trend through the 1987-1995 period.[5] The end point of the trend is still within the expected range of the generic values in NUREG/CR-4550.

The MDP mean probabilities of failure on demand used in plant-specific IPE studies were compared with the results of this study. The PWR IPE mean values were generally consistent with the results of this study and NUREG/CR-4550. The IPE mean values for BWR RI systems were also consistent with the results of this study and NUREG/CR-4550, except for the RHR and RBCC systems. Sixty percent of the IPE mean values for the BWR RHR system were higher than the RHR system value range estimated in this study. Most (approximately 89%) of the IPE mean values for the RBCC system were higher than the RBCC system value range estimated in this study.

[5] Due to change over from the industry's Nuclear Plant Reliability Data System (NRDS) to the Equipment Performance Information Exchange System (EPIX) there was insufficient data available for trending purposes beyond 1995 for the TDP, MDP, and AOV studies. Data on failures and demands from ESF actuations was available through the entire 1987-1998 period.

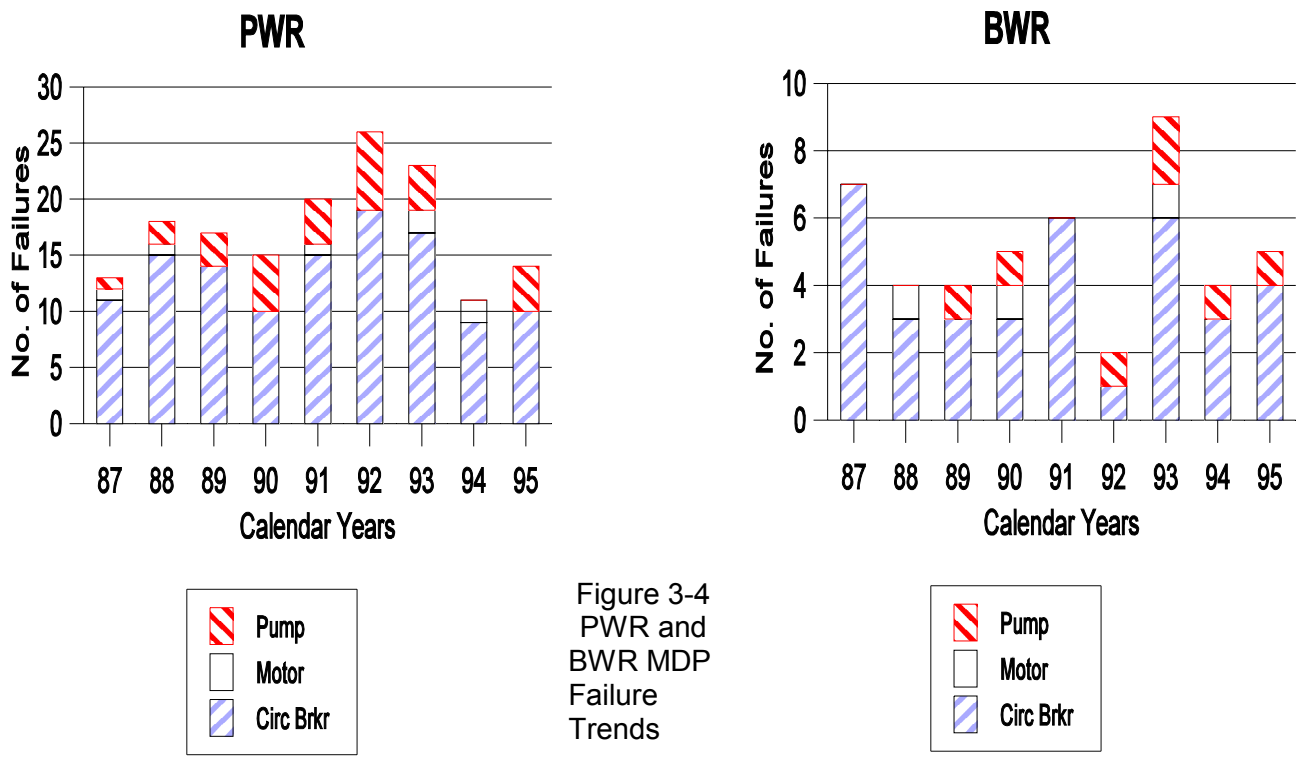

Figure 3-4 PWR and BWR MDP Failure Trends

Failure trends for the PWR and BWR RI systems during the 1987-1995 period were relatively constant. Figure 3-4 shows the MDP failure trends for the 1987-1995 period.

Failure rates, as a function of component-years, were compared among the PWR and BWR plant age groups (three groups, of approximately equal size, from older to newer plants by commercial operations date). For PWRs and BWRs, the review of plant age groups did not show evidence of an increase in MDP failure rates for any of the plant age groups due to aging mechanisms.

The evaluation of MDP subcomponent failure patterns demonstrated that circuit breakers were significant contributors to the MDP failures in both PWR and BWR risk important systems (greater than 75%).

Failures of MDP assemblies in PWR risk important systems were mainly attributed to unknown causes (43%) because a root cause analysis was generally not performed on the predominant failed

subcomponent (circuit breaker). Age/wear and maintenance/procedural deficiencies together accounted for the bulk of the balance (43%). For BWR RI systems, age/wear was the predominant cause (43%), while unknown causes and maintenance/procedural deficiencies together accounted for the bulk of the balance (46%).

3.4 Component Performance Study - Air-Operated Valves, 1987-1998, NUREG-1715, Vol. 3

This study provides the performance evaluation based on industry experience during the 1987 through 1998 period for air-operated valves (AOVs) in the PWR and in the BWR risk important systems.

For the PWR and BWR risk important systems, the AOV probability of failure on demand estimates were consistent with the generic values from NUREG/CR-4550 (used as an input to NUREG-1150), although the PWR RHR system mean value (5.2E-4) is about a factor of 4 lower than the generic mean value (2E-3). Table 3.3 lists the probability of failure on demand estimates developed for the risk important (RI) systems selected for this study and the NUREG/CR-4550 values.

TABLE 3.3
AOV PROBABILITY OF FAILURE ON DEMAND (1987-1998)

	LOWER BOUND	MEAN	UPPER BOUND
NUREG/CR-4550	**5.4E-4**	**2E-3**	**4.8E-3**
PWR RI SYSTEMS:			
Auxiliary Feedwater (AFW)	4.6E-6	1.8E-3	6.9E-3
High Pressure Injection (HPI)	4.8E-6	1.2E-3	4.7E-3
Residual Heat Removal (RHR)	6.1E-5	5.2E-4	1.3E-3
Chemical and Volume Control (CVCS)	3.5E-7	3.4E-3	1.5E-2
Component Cooling Water (CCW)	6.7E-5	5.8E-3	2.1E-2
BWR RI SYSTEMS:			
Reactor Core Isolation Cooling (RCIC)	3.5E-4	3.0E-3	7.7E-3
High Pressure Coolant Injection (HPCI)	4.3E-4	3.6E-3	9.5E-3
Low Pressure Core Spray (LPCS)	2.9E-15	2.1E-3	1.2E-2

The AOV mean probabilities of failure on demand used in plant-specific IPE studies were compared with the results of this study. The PWR IPE mean values were generally consistent with the results of this study and the NUREG/CR-4550 generic values. No comparison was made with BWR IPE mean values as few BWR plant IPEs provided AOV failure probabilities on demand.

For the PWR RI systems during the 1987-1995 period[6], there was a decreasing failure trend. For BWR RI systems no trending was determined due to sparsity of failure data. Figure 3-5 shows the PWR AOV failure trends for the 1987-1995 period.

Failure rates, as a function of component-years, showed no significant variance among the PWR plant age groups (three groups, of approximately equal size, from older to newer plants by commercial

[6] Due to change over from the industry's Nuclear Plant Reliability Data System (NRDS) to the Equipment Performance Information Exchange System (EPIX) there was insufficient data available for trending purposes beyond 1995 for the TDP, MDP, and AOV studies. Data on failures and demands from ESF actuations was available through the entire 1987-1998 period.

operations date). The review of PWR plant age groups did not show evidence of an increase in failure rates for any plant age groups due to aging mechanisms. For BWRs, failure data was too sparse for trending failure rates.

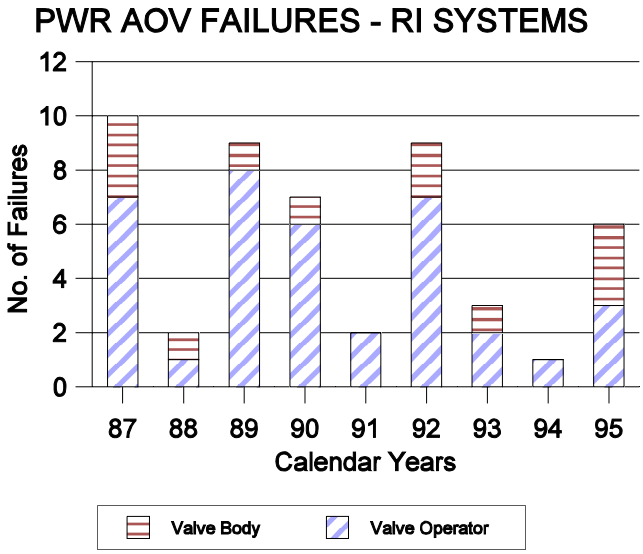

Figure 3-5
PWR AOV Failure Trends

The number of complete AOV common-cause (CCF) failures identified in this study (one) is consistent with the expected number based on the CCF database parameters for the combined PWR and BWR complete failure population (55) used in this study.

The AOVs have two subcomponents (valve body and valve operator). The valve operator was the biggest contributor to AOV failures (76%). Although BWR AOVs also showed that valve operators were the biggest contributors, the number of failures (6) was too sparse to use. Failures of AOV assemblies in PWR RI systems were mainly due to age/wear causes (47%).

4. CASE STUDIES

4.1 Assessment of Risk Significance of Issues at D.C. Cook Nuclear Power Plant, NUREG-1728

This report documents the assessment of the risk significance of 141 issues identified at the Donald C. Cook Nuclear Plant, Units 1 and 2 (Cook 1 and 2) since August 1997. This assessment is part of the agency's Accident Sequence Precursor (ASP) Program. The risk significance was evaluated for each individual issue and the combined issues for the "as-found" conditions at Cook 1 and 2 in August 1997.

Changes in risk resulting from modifications implemented by the licensee were not incorporated into the assessments provided in this study[7].

The evaluation reviewed the following sources of operational experience to identify issues: all LERs issued for Cook 1 and 2 between August 1997 and December 1999; NRC inspection reports for Cook 1 and 2 issued since August 1997; several condition reports that pertained to open (as of 12/28/99) operability evaluations of risk-significant systems; and several licensee self-assessment reports.

Of the 141 issues analyzed, four issues had estimated changes to the baseline core damage frequencies (ΔCDF) greater than 1.0×10^{-6}/year, which qualified them as accident sequence precursors. Each of the four precursors involved conditions that existed since Cook 1 and 2 received their operating licenses. The cumulative ΔCDF resulting from all issues identified at Cook 1 and 2 was estimated to be approximately 5.1×10^{-4}/year for each of the units. High-energy line break (HELB) issues, degraded capability of some equipment to withstand seismic events, and potential pressure locking conditions in two motor-operated valves were the dominant contributors to the CDF increase. The four precursors included two postulated HELB scenarios that contributed ΔCDFs of 4.3×10^{-4}/year and 3.3×10^{-6}/year, respectively; the CDF increase attributed to the degraded seismic capacity that was 3.2×10^{-5}/year; and the sum of small, medium and large loss-of-coolant accident (LOCA) sequences associated with the potential pressure locking conditions in the motor-operated valves that contributed a ΔCDF of 3.7×10^{-5}/year. The risk significance of the combined impact of containment-related issues was small. The effect of the 141 issues was evaluated in terms of their respective impact on the ΔCDF for the initiating events. The resulting ΔCDF for the individual initiating events are summarized in Table 4.1.

Table 4.1: Summary of ΔCDF Results for Individual Initiating Events, D.C. Cook

	Initiating Event	ΔCDF (per year)
1.	High-Energy Line Breaks (HELBs): HELBs in turbine building - HELB in pipe chase adjoining CCW pump room - Unit 2 only	4.3×10^{-4} (total) 4.3×10^{-4} 3.3×10^{-6}
2.	Seismic Events: Failed ESW backwash capability - Failed EDG - Degraded PORV and AFW -	3.2×10^{-5} (total) 3.2×10^{-5} 3.1×10^{-8} 3.2×10^{-7}
3.	Medium LOCA: pressure locking of sump valves	3.2×10^{-5}
4.	Large LOCA: pressure locking of sump valves	4.0×10^{-6}
5.	Loss of offsite power (LOSP)	5.1×10^{-7}
6.	Small LOCA: Pressure locking of sump valves - Other conditions -	 1.4×10^{-6} 2.6×10^{-7}
7.	Intersystem LOCA (ISLOCA)	2.0×10^{-7}

[7]Modifications significantly reduced risk prior to restart.

8.	Transient [excluded LOCAs and LOSPs, and included loss of power conversion system (PCS)]	3.2×10^{-8}
9.	Steam generator tube rupture	1.0×10^{-8}
10.	Loss of control room ventilation	7.7×10^{-10}
11.	Anticipated transient without scram	negligible
12.	Fire Event	negligible
13.	Shutdown Event	negligible
14.	Spent fuel pool event	negligible
15.	Loss of dc power	negligible
Total from all issues		**5.1×10^{-4}**

The following provides the risk and programmatic context for the significance of the study findings:

- Number of accident sequence precursors for Cook 1 and 2 within the 1989-1999 time frame compared to the number of accident sequence precursors in the industry during that time frame - Between 1989-1999, six precursors involving Cook 1 and six precursors involving Cook 2 were identified by the ASP Program. These include the four precursors identified in this study. Approximately 20 percent of all operating nuclear plants have experienced six or more accident sequence precursors over this period.

- Percentage of significant design basis issues (issues whose ΔCDF exceeds $1\text{x}10^{-6}$/year) found at Cook 1 and 2 in August 1997 compared to the percentage in the industry - A recent study of design basis issues (NUREG-1275, Volume 14,"Causes and Significance of Design Basis Issues at U.S. Nuclear Power Plants," November 2000) showed that in 1998 about 1 percent of the design basis issues reported in LERs were accident sequence precursors. Approximately 3 percent of the design basis issues identified at Cook 1 and 2 in August 1997 were accident sequence precursors.

- Risk significance associated with the issues identified at Cook 1 and 2 since August 1997 compared to the results of the licensee's Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) - According to the licensee's IPE and IPEEE, the baseline CDF is approximately 7.8×10^{-5}/year for each of the Cook units. This includes contributions from fire and seismic events as well as internal events. The cumulative ΔCDF resulting from all issues identified at Cook 1 and 2 in this study (5.1×10^{-4}/year), dominated by the four precursors, was approximately 6.5 times greater than the IPE baseline CDF.

- Risk significance of issues identified at Cook 1 and 2 since August 1997 compared to the performance goals of the NRC's strategic plan (NUREG-1614, Vol. 2, Part 1) - In accordance with the classification used in the ASP Program, the ΔCDF of the HELB relating to loss of auxiliary feedwater and safety-related switchgear is an "Important Precursor" since it exceeded 1.0×10^{-4}/year. The cumulative ΔCDF (5.1×10^{-4}/year) for the four precursors (all of which involved design deficiencies that had been corrected before the recent restart) was

approximately 50 percent of the threshold of 1.0×10^{-3} for a “Significant Precursor,“ defined in the performance goals of the NRC’s strategic plan.

Postulated HELBs and conditions similar to those that were precursors at Cook may exist at other plants. Therefore, the staff recommended that an Information Notice (IN) be issued. The IN was issued on December 11, 2000 as IN 2000-20, “Potential Loss of Redundant Safety-Related Equipment Because of the Lack of High-Energy Line Break Barriers.”

One of the objectives of the D.C. Cook study was to use the results to benchmark the Significance Determination Process (SDP) of the new Reactor Oversight Process. The study concluded that the SDP screening process can be overly conservative and the licensee’s input on assumptions and problem definition (or modeling of issues) are essential to obtain an appropriate perspective on the risk significance of issues. A feasibility review performed by an Office of Nuclear Reactor Regulation (NRR) task group (documented in SECY-99-007A, March 22, 1999, Attachment 3) came to a similar conclusion.

5. REGULATORY EFFECTIVENESS REVIEWS

The Office of Nuclear Regulatory Research is assessing selected regulations, including the Station Blackout (SBO) Rule and the Anticipated Transient Without Scram (ATWS) Rule, to determine if regulations are achieving the desired outcomes. This initiative is part of an evolving program to make NRC’s regulatory activities and decisions more effective, efficient, and realistic.

These assessments contributed to the NRC’s goal of maintaining safety by documenting the conclusion that the intended outcomes of the SBO and ATWS rules - to reduced risk to plants - has been achieved. To increase public confidence, drafts of these assessments were made publicly available. The final SBO assessment and ATWS assessment reports addressed public comments.

5.1 Station Blackout Rule Effectiveness

To assess the regulatory effectiveness of the SBO rule (10 CFR 50.63), the expectations were established from objective measures as stated in SBO related regulatory documents in the areas of coping capability, risk reduction, emergency diesel generator (EDG) reliability, and value-impact. The outcomes were obtained from realistic information which includes the operating experience and NRC equipment reliability studies based on actual safety performance. Comparison of the expectations to the outcomes showed whether the expectations were achieved. Discrepancies between expectations and outcomes would prompt a review of the related regulatory documents to identify areas that need NRC staff attention.

The report's conclusion is that the SBO rule was effective considering that the risk reduction expectations were achieved, and that industry and NRC costs to implement the SBO rule were reasonable. In implementing the SBO rule, some plants made hardware modifications (e.g., the addition of diesel generator or gas turbine generator power supplies); and plants generally maintained EDG reliability at 0.95 or better, and established SBO coping and recovery procedures. Consequently, the plants have gained SBO coping capability, reduced risk, increased the tolerance to a loss of alternating current offsite or onsite power, and many plants benefitted from the addition of power supplies by improving operating flexibility.

A comparison of the SBO rule expectations to the corresponding outcomes indicates that resolution of the generic issue of SBO was effective as no additional generic actions are warranted and no new generic safety issues have been identified.

To the extent that the NRC staff revises existing regulatory documents to be more risk-informed and performance-based, the staff is incorporating the lessons learned in the improved inspection procedures and regulatory guides to ensure consistent interpretation and use of terms, goals, criteria, and measurements.

5.2 Anticipated Transient Without Scram Rule Effectiveness

To assess the regulatory effectiveness of the ATWS rule (10 CFR 50.62), the expectations were established from objective measures stated in the rule and accompanying regulatory documents and compared to the outcomes in the areas of risk reduction, modifications, and value-impact. The outcomes were obtained from operating experience after issuance of the ATWS rule. The value-impact assessment determined whether the industry's costs to implement the ATWS rule were reasonable.

The assessment concludes that the ATWS rule has been effective in reducing ATWS risk and the cost of implementing the rule was reasonable. However, uncertainties in reactor protection system (RPS) reliability and mitigative capability warrant continued attention consistent with NRC performance goals to maintain the expected levels of reactor safety and improve NRC effectiveness.

Although past data indicates that the risk from ATWS is in the range foreseen when the ATWS rule was issued, several emerging issues have the potential to impact past achievements. Attention to these issues and compliance with current regulations will assure that the risk from ATWS remains acceptable. These issues are:

- Changes in fuel management strategies and higher fuel burnups have resulted in previously unpredicted oxide growth and fuel assembly distortion, which have resulted in instances of slow

and or incomplete control rod insertion. NRC RPS reliability studies have identified the control rod mechanism as an important common cause failure risk contributor.

- ATWS mitigation on a PWR is highly dependent on the negative moderator temperature coefficient (MTC). Current trends in fuel design are toward longer fuel cycles that results in less negative MTCs at full power for a longer fraction of the cycle time. During this time period, the ATWS mitigation is less effective. Future analyses for PWR reactor licensees should take into consideration the impact of the longer fuel cycle and the resulting impact of the MTCs.

- ATWS mitigation on a BWR is highly dependent on operator actions. Although reasonable human error rates were assumed in the regulatory analyses, it is difficult to confirm that these human error rates have been achieved for this rare event.